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SOUTHERN COMPANY
Energy to Serve Your World℠

April 29, 2004

Docket No.: 50-348

NL-04-0702

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant – Unit 1
Licensee Event Report 2004-001-00
Reactor Trip Due to Steam Generator Feedwater Pump Speed Control Failure

Ladies and Gentlemen:

Joseph M. Farley Nuclear Plant – Licensee Event Report (LER) No. 2004-001-00 is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A).

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

L. M. Stinson

L. M. Stinson

LMS/WAS/sdl

Enclosure: Licensee Event Report 2004-001-00

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., Executive Vice President
Mr. D. E. Grissette, General Manager – Plant Farley
RTYPE: CFA04.054; LC# 14021

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. E. Peters, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

IE22

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request. 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4) Reactor Trip Due to Steam Generator Feedwater Pump Speed Control Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
3	1	2004	2004	- 001 -	00	04	29	2004	FACILITY NAME	DOCKET NUMBER

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

OPERATING MODE (9)	1								
POWER LEVEL (10)	100								
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
			20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. E. Grissette, General Manager Nuclear Plant	334-899-5156

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JB	SIC	W120	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On March 1, 2004 at 0522, with the reactor at 100% power, Unit 1 automatically tripped due to Steam Generator 1C level reaching its high level setpoint. At approximately 0521, the lead-lag card in the Steam Generator Feedwater Pump (SGFP) master speed control circuit failed, causing a ramp increase in the speed of both SGFPs. The speed increase caused an increase in feedwater flow to all three steam generators. As a result of the increase in SGFP speed, a low SGFP suction pressure condition occurred. Operators responded to the low suction pressure condition and the increasing feedwater flow by starting a stand-by condensate pump and manually closing the Main Feedwater Regulating Valves (MFRVs). The MFRVs responded to the transient, but the integrated reactor operator and system response was not rapid enough to prevent steam generator level reaching the high level setpoint. Per design, the main turbine, both SGFPs, and the reactor tripped. All safety systems functioned as designed.

This event was caused by failure of the Lead-Lag (NLL) card in the master SGFP speed control circuit, resulting in a ramp increase in SGFP speed. The unexpected SGFP low suction pressure alarm delayed the operators' diagnosis of the failure, thus delaying operator action which might have prevented the trip.

The failed lead-lag (NLL) card has been replaced. Annunciator response procedures have been revised to provide additional guidance for malfunctions of the feedwater control system.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Westinghouse -- Pressurized Water Reactor
Energy Industry Identification Codes are identified in the text as [XX]

Description of Event

On March 1, 2004 at 0522, with the reactor at 100% power, Unit 1 automatically tripped due to Steam Generator 1C level reaching its high level trip setpoint. At approximately 0521, the lead-lag card in the Steam Generator Feedwater Pump (SGFP)[SJ] master speed control circuit [JB] failed, causing a ramp increase in the speed of both SGFPs. The speed increase caused an increase in feedwater flow to all three steam generators. As a result of the increase in SGFP speed, a low SGFP suction pressure condition occurred.

The first Main Control Board Annunciator to alarm was the SGFP Suction Pressure Low alarm. The operators responded to this alarm per procedure by starting the stand-by Condensate Pump. The feedwater flow increase caused steam generator levels to rise resulting in Steam Generator Level Deviation alarms. After verifying that SGFP suction pressure was being restored, the operators took manual control of the Main Feedwater Regulating Valves (MFRVs). The MFRVs responded to the transient but the integrated reactor operator and system response was not rapid enough to prevent steam generator level reaching the high level setpoint. Due to differences in MFRV position and setup, the 1B MFRV began reducing flow to 1B SG first, diverting additional feed flow to 1A and 1C SGs. When the 1C SG level reached its high setpoint, per design, the main turbine, both SGFPs, and the reactor tripped. All safety systems functioned as designed following the trip.

The SGFP Suction Pressure Low alarm is a "yellow" alarm requiring higher priority response than the steam generator level deviation alarms. The occurrence of the SGFP suction pressure low alarm delayed the operator diagnosis of the failure, thus delaying possible optimum operator action which might have prevented the trip.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Cause of Event

This event was caused by failure of the Lead-Lag (NLL) card in the master SGFP speed control circuit, resulting in a ramp increase in SGFP speed. The card failure was due to aging.

Operator response to the unexpected SGFP low suction pressure alarm delayed the operators' diagnosis of the failure, thus delaying possible optimum operator action which might have prevented the trip.

Safety Assessment

All safety systems functioned as designed after the trip. The main feedwater pumps were recoverable from the control room if they had been needed. Therefore, the health and safety of the public were unaffected by this event.

This event does not represent a Safety System Functional Failure.

This event represents a Reactor Trip with Loss of Normal Heat Removal (LONHR).

Corrective Action

The failed lead-lag (NLL) card has been replaced.

Annunciator response procedures have been revised to provide additional guidance for malfunctions in the feedwater control system.

LICENSEE EVENT REPORT (LER)

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Additional Information

A program to address generic aging issues in the plant process control system (7300 system) is currently being implemented.

The following LERs have been submitted in recent years on Steam Generator Feedwater Pump associated events:

LER 2000-004-00 Unit 2 Reactor Trip Due to Degraded Main Feedwater Regulating Valve Transient Response.

LER 2002-004-00 Unit 1 Manual Reactor Trip on Loss of Both Steam Generator Feed Pumps.

LER 2003-003-00 Unit 1 Unplanned Auxiliary Feedwater Actuation upon Trip of Steam Generator Feed Pump.